ACCESSION #: 9603010227
LICENSEE EVENT REPORT (LER)

FACILITY NAME: James A. FitzPatrick Nuclear Power Plant PAGE: 1 OF 17

DOCKET NUMBER: 05000333

TITLE: Low Reactor Water Level Scram Due to Feedwater Transient
EVENT DATE: 04/20/93 LER #: 93-009-03 REPORT DATE: 02/22/96

OTHER FACILITIES INVOLVED: DOCKET NO: 05000

OPERATING MODE: N POWER LEVEL: 100

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR SECTION:
50.73(a)(2)(i), 50.73(a)(2)(iv), 50.73(a)(2)(v)

LICENSEE CONTACT FOR THIS LER:
NAME: Mr. Gordon Brownell, Senior TELEPHONE: (315) 349-6360
Licensing Engineer

COMPONENT FAILURE DESCRIPTION:
CAUSE: SYSTEM: COMPONENT: MANUFACTURER:
REPORTABLE NPRDS:

SUPPLEMENTAL REPORT EXPECTED: NO

ABSTRACT:

Update report: Previous reports dated 10/25/94, 12/06/93, and 05/19/93. On 4/20/93 at 1654, while operating at 100 percent power, abrupt reduction in feedwater [SJ] flow resulted in rapidly decreasing Reactor water level. A Reactor Scram occurred at 177 inches above top active fuel. Plant operators attempted to restore reactor water level by taking manual control of the reactor feed pumps [JK], but were unsuccessful due to component failures. The event was caused by a loose electrical connection on the A Reactor Feed Pump [JK] Turbine Governor Control. The transient was compounded by a failed discharge check valve [SJ] on Reactor Feed Pump A. These failures allowed feedwater flow to be reduced and flow to be diverted causing a scram on low Reactor water level. All control rods fully inserted on the scram and plant systems responded as designed to decreasing and then increasing reactor water level. A Post Trip Evaluation was conducted to determine the root cause of the event and recommend corrective actions to prevent recurrence: The evaluation identified that the Technical Specification Pressure - temperature

cooldown and heatup limit of 100 degrees per hour had been exceeded. All necessary repairs and corrective actions identified through the evaluation were completed prior to plant restart.

END OF ABSTRACT

Update report: Previous reports date 10/25/94, 12/06/93, and 05/19/93.
EIIS Codes are in []

Event Description

On April 20, 1993, the plant was operating at 100 percent power with Reactor water level being automatically maintained at the normal level of approximately 202 inches above the top of active fuel (TAF). At 1654 hours the Reactor Pressure Vessel (RPV) high/low level annunciator [IB] alarmed. Concurrently the Nuclear Control Operator noted a rapid lowering reactor water level. He immediately took manual control of both Reactor Feed Pumps (RFPs) [JK] and attempted to restore reactor water level. The Reactor water level decrease was halted and then began to trend upward to approximately 187 inches TAF and then rapidly trended down, initiating a reactor scram on Reactor Low Level (177 inches above TAF). Plant operators carried out the actions of Abnormal Operating Procedure (AOP), Reactor Scram, and Emergency Operating Procedure (EOP), RPV Control. EOP-2 was exited at 1721 hours with the plant stable in a hot shutdown condition. The cause of rapidly lowering Reactor Water Level was an intermittent control signal failure to Reactor Feed Pump Turbine A motor gear unit compounded by a concurrent failure of Reactor Feed Pump A discharge check valve [SJ].

The sequence of events leading up to and immediately following the scram is presented below:

April 20, 1993:

1654:00 Hours: - Normal plant operation at 100 percent rated power. No evolutions, planned or in progress with the potential to impact plant operation. Both RFP turbines operating normally in automatic for control of Reactor Level at approximately 202 inches above TAF.

1654:17: - Reactor Water Level lowers to the low level alarm at 196.5 inches TAF. Reactor Water Recirculation [AD] Pump runback initiation was observed by the engineer on shift.

1654:25: - Runback of both Reactor Water Recirculation
Pumps to the #2 speed limiter (44 percent
speed).

1654:36: - Reactor Scram on Low RPV Level (177 inches
above TAF).

1654:44: - High Pressure Coolant Injection [BJ] (HPCI)
initiation at a recorded level of approximately
130 inches above TAY.

1654:48: - Reactor core Isolation cooling [BN] (RCIC)
initiation at a recorded level of approximately
128.4 inches above TAF.

- Alternate Rod Insertion [JD] (ARI) initiation at
a level of approximately 128.4 inches
above TAF.

- Both Reactor Water Recirculation Pumps trip at
approximately 128.4 inches above TAF.

- RPV water level at 128.4 inches TAF (lowest
recorded level).

1655:06 and 1655:14 - B Reactor Feed Pump bearing high vibration.

- RPV water level rapidly increasing
(approximately 5 inches per second)

1655:22 - RPV water level reaches level 8 (222.5 inches
TAF).

- Main Turbine [TA] Trip.

- RCIC Trip.

- RFPs Trip.

- NOTE: Post Trip Review indicated that HPCI
also tripped on high Reactor water
level, however, this trip was not

recorded.

1656 - Plant operators completed immediate actions of AOP-1, Reactor Scram. All control rods verified to be fully inserted.

1704 - RPV water level decreases below level 8 (222.5 inches TAF). (Highest recorded RPV level on transient was 230.8 inches TAF).

1710 - Reactor Feed Pump A manually started. Low Flow Feedwater Control [JK] valve placed in operation.

1712 - Reactor Scram reset.

1714 - Plant Process computer [ID] program OD-7 shows all control rods fully inserted.

- Alternate Rod Insertion (ARI) reset.

1721 - Plant Operators exited Emergency Operating Procedure No. 2 (EOP-2).

1725 - Plant Operator noted RFP A speed oscillating approximately 400 RPM with controller in manual.

- Nuclear Control Operator secured the B Condensate [SG] and the B Condensate Booster [SG] Pump.

1726 - Plant Operator started the B Reactor Feed Pump in preparation to securing the A Reactor Feed Pump.

1732 - The A Reactor Feed Pump bearing vibration alarm recorded.

- The A Reactor Feed Pump motor gear unit indication was observed to be blinking on and off.

1735 - Plant Operator began restoration of High

Pressure Coolant Injection to a standby lineup. Upon reset of the High RPV water level trip, HPCI was inadvertently started due to a procedure deficiency.

The Operator closed the HPCI steam supply valve, 23MOV-14, and tripped the HPCI Turbine at approximately 500 RPM and prior to injection to the RPV.

1742 - The A Reactor Feed Pump was secured.

1744 - Reactor Feed Pump A speed indicated 3400 RPM on the local indicator. (Subsequently determined to be rotating backward due to failed discharge check valve.)

- High Pressure Coolant Injection system reset and placed into a standby lineup.

1745 - Reactor Core Isolation Cooling system reset and placed into a standby lineup.

1746 - Group II, primary containment isolations [JM] reset.

1749 - Reactor Building [NG] ventilation restored to normal lineup.

- ENS notification performed.

1755 - Entered Limiting Condition for Operation due to apparent failure of drywell floor drain containment isolation valve, 20MOV-82 to fully close.

1810 - Closed Reactor Feedwater Pump A inlet to FCV-137 Isolation Valve, 34FWS-16A, due to pump speed indicating 3000 RPM after the pump was secured. The pump then coasted to a stop.

1815 - Reactor Feed Pump A on the turning gear.

1830 - 125VDC ground on the B station battery [EJ].

Entered AOP-23, DC Power System B Ground
Isolation Procedure.

1921 - Ground isolated to control circuitry for the High Pressure Coolant Injection Steam Supply valve, 23MOV-14. Upon re-closing the breaker for 23MOV-14 the ground was clear.

1935 - Commenced Reactor cooldown.

2005 - Ground returned on the B Station Battery. opened breaker for 23MOV-14 and ground cleared. Declared High Pressure Coolant Injection System inoperable and entered Limiting Condition for Operation. (This event is further described in LER-93-010).

2130 - ENS notification for HPCI inoperability completed.

Cause of Event

The plant was operating at 100 percent power when a control signal failure of the A Reactor Feed Pump motor gear unit caused the A Reactor Feed Pump speed to lower. This resulted in a steam-feed flow mismatch which caused RPV level to rapidly lower to the low level alarm (196.5 inches TAF). Upon receipt of the low level alarm, the Nuclear Control Operator placed feedwater control in manual and attempted to recover RPV level; however, this was unsuccessful due to the nature of the failure of the discharge check valve on Reactor Feed Pump A.

The RFP speed control transient coupled with a failed Reactor Feed Pump A discharge check valve caused Reactor Feed Pump A flow to stall due to the inability to develop the required discharge head at the reduced speed. When Reactor Feed Pump A flow lowered to less than 2500 gallons per minute, the following automatic actions occurred:

1. Reactor Feed Pump A minimum flow valve opened.

2. Reactor Recirculation Pump runback to the Number 2 Speed Limiter was initiated (RPV level 196.5 inches TAF and less than two Reactor Feed Pumps running).

Reactor Power began to lower as a result of the Recirculation Pump runback; however, when the A Reactor Feed Pump minimum flow valve opened an additional approximately 1 X 10E6 pounds mass/hour feedwater flow was diverted from the feedwater system back through the failed check valve to the main condenser. This diversion of feedwater flow from the RPV resulted in a rapidly lowering RPV level and a reactor scram at a level of 177 inches above TAF. The power reduction from the successful scram was sufficient for the B Reactor Feed Pump, assisted by HPCI and RCIC, to recover reactor water level from a low level of 128.4 inches to 230.8 inches in approximately 34 seconds.

The root cause of this event was:

1. Control signal failure of the A Reactor Feed Pump motor gear unit due to a loose electrical connection. A terminal screw was found stripped in the controller terminal block.

2. Failure of the A Reactor Feed Pump discharge check valve, occurred at some time prior to the scram, due to turbulent flow within the valve which resulted in fretting between the anti-rotation pin and the swing arm and ultimately gross failure.

Analysis of Event

The plant responded as designed to the decreasing feedwater event. A post trip evaluation of the event was completed in order to evaluate the plant response and the operator actions taken in response to the transient. On-shift operators responded to the transient in an expedient and professional manner using all appropriate plant procedures. The following deficiencies were identified by plant operators during the transient recovery:

1. The High Pressure Coolant Injection System was inadvertently started (with no injection to the RPV) while restoring the system to a standby lineup due to a procedural deficiency.

2. Valve 20MOV-82, drywell floor drain [WD] pump discharge Primary Containment [NH] isolation valve failed to indicate fully closed upon receipt of a Group II isolation signal [JA]. The Isolation was promptly accomplished by closing of 20AOV-83, the second containment isolation valve.

3. The High Pressure Coolant Injection System automatic design

response (initiate and inject at rated flow within 30 seconds) was not readily verified by the operating crew due to insufficient data and the short duration of the transient.

4. The High Pressure Coolant Injection System was declared inoperable, subsequent to the transient, upon discovery of a 125 VDC ground on the steam supply valve, 23MOV-14 (refer to LER-93- 010).

During the post trip evaluation it was identified that the cooldown limit of 100 degrees per hour had been unintentionally exceeded on the lower RPV head. The thermal transient was evaluated and determined to be within the original design basis, bounded by the loss of feedwater pump transient evaluated in the original FitzPatrick RPV stress report. Correspondence with General Electric supports this evaluation. The 100 degree per hour heatup, rate was unintentionally exceeded, several hours following the transient, probably as a result of reactor vessel natural circulation flow development. Forced recirculation had not yet been restored at that time.

The safety significance of the event was minimal because the plant responded as designed to a lowering RPV water level. Operator actions during the event were timely and appropriate. RPV pressure remained stable throughout the event with the Main Turbine Electrohydraulic Control System [TG] controlling as designed. No safety relief valves [SB] were actuated or challenged. There was no challenge to the primary containment [NH).

The event requires a report under 10 CFR 50.73(a)(2)(iv). That is, a condition that resulted in automatic actuation of the Reactor Protection System [JC]. The event also requires a report under 10 CFR 50.73(a)(2)(v)(D). That is, a condition that could have prevented the fulfillment of the High Pressure Coolant Injection System that is needed to mitigate the consequences of an accident.

The event also requires a report under 10 CFR 50.73(a)(2)(i)(B). That is, the non-nuclear heatup, and cooldown limits specified in Technical Specification 3.6.A.3 were exceeded for the RPV bottom head and the incomplete logic testing of the HPCI Injection Valve, 23MOV-19, required by Technical Specification Table 4.2.2.

Corrective Action

The Post Trip Evaluation identified the following actions or issues requiring resolution. Post restart corrective actions identified as a result of the evaluation are annotated with assigned due dates.

A. An initial review of available data indicated that the High Pressure Coolant Injection System (HPCI) may not have injected as designed at rated flow during the transient. Engineering conducted a detailed investigation in evaluating the HPCI response including the following:

1. Station battery B ground affect on the Turbine Steam Admission valve, 23MOV-14, or the HPCI control system, or the interlock to open pump discharge valve, 23MOV-19. The 125VDC ground was due to a pinched electrical wire shorted to ground on the Limitorque motor for 23MOV-14. The location of the ground would not have affected the 125 VDC battery, the operation of the 23MOV-14, or the 23MOV-14 interlock with the 23MOV-19 valve. This failure is further described in LER-93-010.

2. The HPCI control system parameters (voltages and currents) were verified normal. A non-intrusive check of the hydraulic system showed no abnormalities. No hydraulic leaks were observed.

3. A review of the HPCI initiation logic determined that the logic was functional; however, this portion of the investigation revealed that the routinely performed logic testing uses a GE test switch. For the 23MOV-19 valve the GE test switch only tests one side (high drywell pressure) of the logic. The low RPV level side was not tested. Based upon this apparent logic test omission, this is a violation of Technical Specification Table 4.2.2 requirements. LER-92-032 described a similar failure to test all features of a logic path. Surveillance Test Program Improvement is included in the FitzPatrick Results Improvement Program and is designed to ensure that all logic paths and components are properly tested. Two subsequent action items were opened:

a) ACTS 7820, Assigned the need to critique the HPCI logic test omission, identify other logic function test surveillance that uses the GE test switch, and to review whether the switch tests all logic paths in a sample (approximately 25 percent) of the identified logic tests. Complete, October 26, 1993.

b) ACTS 7821, Perform a detailed review of the HPCI logic test. Complete, June 1, 1993.

4. HPCI pump discharge valves 23MOV-19 and 23HPI-18 were determined to have functioned properly.

5. HPCI pump discharge flow transmitter, 23FT-82, was verified to be calibrated within tolerance and no air was observed when venting was performed.

6. The process computer post trip log [ID] (EPIC) Feedwater header pressure was analyzed for the time period during the transient that the HPCI system would be required to have injected. It was determined that no adverse interaction between feedwater and HPCI existed. A volumetric flow analysis was performed (calculation, JAF-CALC-Misc-99001) which demonstrated that the HPCI system injected as required.

The Engineering evaluation concluded that all HPCI related equipment functioned properly as designed during the transient. In order to further demonstrate HPCI system operability, the evaluation recommended performing HPCI Surveillance Test prior to exceeding 150 psig and again at >900 psig. This item was completed during the startup power ascension plan.

B. Inspect Reactor Feed Pump Discharge Check Valves, 34FWS-4A and 4B. The Maintenance Department conducted an equipment failure evaluation of the A Reactor Feed Pump discharge check valve in addition to an inspection of the B discharge check valve. The root cause of the A Reactor Feed Pump check valve failure appears to be design errors in the pump and piping configuration and valve application. Flow induced vibration has been cited as the cause of repeated valve failures.

An Engineering evaluation of the valve 34FWS-4A was performed in 1989 as a result of an extremely short 13 month life cycle. This evaluation recommended adding a "foot" to the swing arm end and welding the hanger bracket bolts to the hanger bracket. These modifications were effective in reducing hanger attachment bolt failures, but did not prevent wear on the disc stud and swing arm which was the cause of the current failure.

A preventative maintenance evaluation of the valve 34FWS-4A issued in April, 1990, recommended valve inspection every two years. Valve 34FWS-4A was last inspected in May, 1992. Check valve 34FWS-4B appears to be less susceptible to similar failure since its piping configuration results in lower flow turbulence than that existing at the 34FWS-4A valve.

Based upon the check valve equipment failure evaluation, three action items were opened:

a) ACTS 7621, Disassemble and inspect 34FWS-4A during the Fall, 1993 Maintenance Outage. Complete, November 16, 1993.

b) ACTS 7622, Evaluate potential for damage to 34FWS-4A/B during long path feedwater recirculation operation. Complete, September 1, 1993.

c) ACTS 7558, Add modification F1-89-151 (replace check valves 34FWS-4A and B) to the 1995 refuel outage scope. Status: The Reactor Feedwater Pump A discharge check valve was replaced, as scheduled, during the 1994 Refuel Outage, with an improved nozzle type valve design. The valve has operated successfully through two plant shutdowns and three plant startups since its installation. Results of inspections conducted on Reactor Feedwater Pump B discharge check valve during the 1994 Outage, and subsequent reliable valve performance data, have demonstrated and provided assurance that conditions do not exist or warrant the need to replace the discharge check valve during the 1996 Refuel Outage. However, the Reactor Feedwater Pump B discharge check valve will continue to be closely monitored, including a valve inspection each refuel outage to assure optimum valve performance is maintained.

C. The review of data from the plant transient indicated a probable feedwater level control malfunction. The Instrument and Control Group performed troubleshooting based upon data gathered during the event and upon operator observations. Troubleshooting identified that the A Reactor Feed Pump motor gear unit was driving the turbine control into the low speed stop and then trying to return to a normal position on a frequently recurring cycle. The cause was determined to be a loose electrical connection due to a stripped terminal screw. The terminal strip was replaced and all other connections verified tight.

D. The Maintenance, Performance Engineering, and operations staff evaluated the transient effect on both Reactor Feed Pumps A and B. Vibration data gathered during the event was reviewed. Confirmation that the A Reactor Feed Pump rotated backwards was determined by reviewing thrust bearing temperature data recorded during the event.

The staff determined that the A Reactor Feed Pump was rotated backwards due to reverse flow through the pump while the turbine was still coupled. Rotation at approximately 3000 RPM was observed at the local speed indicator at the same time as temperature shifts on the thrust bearing were recorded which confirmed reverse rotation.

Performance Engineering reviewed data obtained during the event by the plant process computer and installed vibration monitoring equipment. The analysis of data showed that no bearing exceeded the steady state operating limit of 160 degrees fahrenheit nor did any vibration readings indicate either pump operated in the danger range during this event. Although both Reactor Feed Pumps experienced significant speed oscillations during the transient, neither pump reached the overspeed setpoint of 5000 revolutions per minute. The vendor manual states that in the event the pump runs backwards due to check valve failure, it will probably not be damaged so long as the bearings are lubricated. The next RFP start should be carefully monitored to be sure it is operating normally. Maintenance personnel reported no evidence of wear products in oil samples taken. An oil sample was sent offsite for analysis. The results indicated light ferrous deposits and normal wear.

Based on the vendor recommendations and staff analysis of data the following action items were opened:

a) ACTS 7529, Operations Department should closely monitor both feed pumps and turbines for adverse vibration or temperature trends during the startup. This item was completed during the startup power ascension.

b) ACTS 7530, Performance Engineering should obtain a complete vibration baseline for both units after startup, in order to benchmark machinery condition. This item was completed on May 10, 1993.

E. Based upon the loose electrical connection found in the A Reactor Feed Pump control circuit, it was determined that all open Work Requests should be reviewed to determine the potential for the existence of other deficiencies due to damaged terminal boards, loose terminals or burnt connections. A total of 1482 work requests were reviewed. Fourteen work requests were identified as needing further investigation. Following walkdown by the responsible work groups, three of the fourteen work requests were identified as requiring additional work prior to startup.

F. During the recovery from the event, while restoring the High Pressure Coolant Injection System to a standby lineup, the turbine was inadvertently started. Operations personnel investigated the cause and determined that the operating procedure did not provide guidance on resetting the High Pressure Coolant Injection System initiation signal with a high RPV water level trip still locked in. one section of the procedure states that with the initiation signal reset the turbine will remain shutdown until an initiation signal occurs. In actuality, if the steam supply valves are open and the oil pump is running, the High Pressure Coolant Injection System will restart when the high RPV level is reset.

Changes were made to the High Pressure Coolant Injection Operating procedure in order to remove the confusing section and to add steps describing how to reset the high level turbine trip signal.

G. During the event a Primary Containment Isolation was initiated and 20MOV-82, Drywell Floor Drain Sump Discharge valve, failed to indicate full-closed. Both the red and green light remained on, indicating that the valve was in an intermediate position. The valve was subsequently found in the fully closed position. The cause of the inaccurate position indication was due to the red (open) light limit switch being set too close to the fully closed position. This was confirmed through valve diagnostic testing which confirmed proper valve operation. Current procedural guidance allows the red light to be adjusted no more than one quarter of a second before disc motion is stopped by seating and before the close torque switch opens. The red light was readjusted within procedural guidelines, yet prior to disc stopped motion.

An action item was opened to initiate a procedure revision:

a) ACTS 7620, revise Maintenance Procedure 59.36, Data Acquisition of Limitorque valves (Votes) to ensure that the red light is set sufficiently far from the fully closed position to preclude dual indication during operation. Complete, July 23, 1993.

H. When evaluating the plant response to the transient, the post trip review team found that the Final Safety Analysis Report (FSAR) description for the loss of feedwater transient assumes a Group I isolation (Main Steam Isolation Valves) at an RPV level of 126.5 inches TAF. The Group I isolation logic was previously modified to initiate isolation at an RPV level of 59.5 inches TAF in April, 1987, but the FSAR was not revised at that time. A Deviation Event Report was initiated to investigate and resolve the discrepancy.

I. Technical Services was tasked to investigate possible RPV bottom head thermal stratification based upon data from this event. This data indicated that the vessel bottom head had been subjected to a cooldown and a heatup cycle which were greater than 100 degrees in an hour. This investigation provided recommendations to eliminate or minimize thermal RPV transients. Based upon this investigation, changes were made to Abnormal Operating Procedure No. 1, Reactor Scram.

Following the scram, the operating crew was actively involved in post scram recovery tasks. Prioritization of these tasks was necessary because certain tasks are more important than others. FitzPatrick procedures for restoration of the Reactor Water Recirculation and Reactor Water Cleanup systems are not written to facilitate rapid recovery of either system which contributed to exceeding the RPV bottom head pressure-temperature limits. Restoration of these systems was given a low priority. A separate critique was completed for this portion of the event. Input from the BWR owners group and industry experience were provided to the personnel that performed the critique. Additional procedure change recommendations have been provided to address this issue. The recommendations were evaluated by management and have been fully implemented as of January 18, 1994.

General Electric was contacted to assist in evaluating the thermal transients. Following the scram, which included shutdown of the recirculation pumps, there was significant cooldown of the bottom head as a result of loss of forced recirculation coupled with the continued operation of the CRD pump. During this cooldown, the pressure-temperature conditions exceeded the limits shown in the Pressure-Temperature curves in the Technical Specifications Figure 3.6-1, Part 1. The action statement for the violation was followed. That is, place the plant in cold shutdown and evaluate the transient prior to restart. The Technical Specification pressure-temperature curve is based upon a combination of the limiting curves corresponding to the 10 CFR 50, Appendix G postulated event for a flaw in the feedwater nozzle and the beltline region.

A pressure-temperature curve specific to the FitzPatrick vessel bottom head was made available by General Electric. Evaluating the transient cooldown against this curve shows that there is no violation of the 10 CFR 50, Appendix G temperature limit for the bottom head.

The original RPV stress analysis dated December 15, 1971, shows that the loss of feedwater transient assumes a temperature gradient decrease from 480 degrees to 100 degrees in 1.2 hours. The cooldown event of April 20, 1993, is within this original design basis and closely approximates the loss of feedwater pump transient evaluated in the original RPV stress report.

The violation of the 100 degree per hour heat-up of the bottom head occurred as the RPV natural circulation thermal driving head developed and progressed to the point where the cooler water was eventually removed from the lower head region. From a fracture mechanics viewpoint, this is not considered significant since the heat-up produces compressive stresses. Furthermore, the 100 degree per hour requirement is not important in terms of the resulting stresses.

Both the cooldown and heatup limit violations were significant from an operational perspective. Although there was recent industry experience in this area which could have been used to minimize, if not prevent the transients, plant operators had not yet been made aware of the initiating conditions or methods to reduce the severity of thermal transients on the bottom RPV head.

Plant Operators were provided the following guidance through procedure changes to AOP-1, Reactor Scram.

1. Reset the scram signal as soon as practicable.

2. Close the recirculation loop discharge valve then throttle open for three seconds.

3. Restore Reactor Water Cleanup System as soon as practicable.

The following post startup action item has been assigned:

a) ACTS 7550, revise AP-03.01, Post Trip Evaluation, to collect heatup/cooldown data and evaluate RPV Pressure-Temperature limitations. Complete, June 1, 1993.

J. Due to questions regarding the operation of the mode switch during the event, Surveillance Test 29B was performed prior to startup in order to verify proper operation.

In addition, a subsequent action item was opened for training:

a) ACTS 7533, revise operator training to ensure that all Operators are aware that the Control Room Reactor Mode Switch is more difficult to re-position than is the Simulator Mode Switch. Complete, June 14, 1994.

K. Lack of easily obtainable transient data led to delays in conducting the post trip evaluation, including confirmation of the High Pressure Coolant Injection System operation. The following parameters have been added to the EPIC (Plant Process Computer) post trip log:

1. Reactor Feed Pumps A and B speed.

2. Level transmitters for High Pressure Coolant Injection and Reactor Core Isolation Cooling High RPV Level Trip.

3. High Pressure Coolant Injection pump speed, discharge pressure, and steam flow.

4. Reactor Core Isolation Cooling pump speed discharge pressure, discharge flow and steam flow.

L. This sequence of events was added to the simulator scenario bank on September 15, 1993.

Similar Previous Events: LER-90-009 describes a similar event in which a loose part in the Feedwater level control circuitry resulted in a false low reactor water level signal and unit trip.

Failed Component Data:

Component: 34FWS-4A, Feedwater System Pump A Discharge
Check Valve

Component Manufacturer: Velan Valve Corporation
18 inch forged bolted bonnet swing check
valve, Model No. B21-2A114B-2TS

Replacement Component Data:

Component: 34FWS-4A, Feedwater System Pump A Discharge
Check Valve

Component Manufacturer: Enertech
18 inch Nozzle Check Valve
Model No. DRV-B

ATTACHMENT TO 9603010227 PAGE 1 OF 2

Attachment 1

LER-93-009-03

Commitment Status

Number Commitment Due Date

JAFP-96-XXXX-01 Reactor Feedwater Pump B Discharge Each
Check Valve Will Continue to Be Refueling
Closely Monitored, Including a Valve Outage
Inspection Each Refuel Outage to Assure
Optimum Valve Performance if Maintained

ATTACHMENT TO 9603010227 PAGE 2 OF 2

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New York Power Harry P. Salmon, Jr.
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February 22, 1996
JAFP-96-0072

United States Nuclear Regulatory Commission
Document Control Desk
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Washington, D.C. 20555

SUBJECT: DOCKET NO. 50-333
LICENSEE EVENT REPORT: 93-009-03

Low Reactor Water Level Scram Due to Feedwater Transient

Dear Sir:

This report was submitted in accordance with 10 CFR 50.73 (a) (2) (iv), 10 CFR 50.73 (a) (2) (v) (D) and 10 CFR 50-73 (a) (2) (i) (B).

This supplement provides an updated status of the long-term corrective actions related to this event. Reactor Feedwater Pump A discharge check valve was replaced, as scheduled, during the 1994 Refuel Outage, with an improved nozzle type valve design. Results of inspections conducted on Reactor Feedwater Pump B discharge check valve during the 1994 Outage, and subsequent reliable valve performance data, have demonstrated and provided assurance that conditions do not exist or warrant the need to replace the discharge check valve during the 1996 Refuel Outage. However, the Reactor Feedwater Pump B discharge check valve will continue to be closely monitored, including a valve inspection each refuel outage to assure optimum valve performance is maintained.

Questions concerning this report may be addressed to Mr. Gordon Brownell (315) 349-6360.

Very truly yours,

HARRY P. SALMON, JR.

HPS:GJB:las
Enclosure

cc: USNRC, Region 1
USNRC Resident Inspector
INPO Records Center

*** END OF DOCUMENT ***
